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ITER Scenario Performance Simulations Assessing Control and Vertical Stability

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Abstract

In simulating reference scenarios proposed for ITER operation, we also explore performance of the poloidal field (PF) and central solenoid (CS) coil systems using a controller to maintain plasma shape and vertical stability during the discharge evolution. We employ a combination of techniques to evaluate system constraints and stability using time-dependent transport simulations of ITER discharges. We have begun the process of benchmarking these simulations with experiments on the DIII-D tokamak. Simulations include startup on the outside limiter, X-point formation and current ramp up to full power, plasma burn conditions at 15MA and 17MA, and ramp down at the end of the pulse. We also simulate perturbative events such as H-to-L back transitions. Our results indicate the viability of proposed ITER operating modes.

1. Simulation techniques

In solving for free-boundary equilibria with thermal transport in the CORSICA [1] code, we use a combination of transport simulation techniques [2] to assess coil-system performance and stability. In fast simulations to explore parameter variations, we use a series of prescribed shapes to evolve the discharge. After each time step, coil currents consistent with the shape evolution are obtained from a subsequent free-boundary solution. For forward control, the ITER controller (JCT-2001, VS1) maintains the plasma via feedback control of the reference shape at the gap (difference between a reference separatrix and the boundary obtained) locations. In this case, vertical stability is provided by the fast feedback loop response controlling the equilibrium centroid velocity. In these forward simulations, only the controller-predicted PF and CS coil voltages are fed back to the equilibrium code to maintain control. The coil currents obtained are sensitive to the transport assumptions used in simulations with the resistive flux evolution altering the overall coil current demands. Using scaling of ITER size and time scale, a series of startup similarity experiments are being conducted on the DIII-D tokamak [3,4] and are providing data to validate the controller transport modeling described here.

In the high auxiliary heating phase (burn), we obtain an H-mode-like pedestal from the transport model with its associated edge bootstrap current peak from a neoclassical model. The presence of this edge current alters the internal inductance ($l_i(3)$ used here) and is important for exploring stability and control system performance. Coil current limits stemming from the magnetic field strength and forces at the conductors are monitored in these simulations to further assess the adequacy of the coil system. During plasma evolution, we calculate the vertical instability growth rates. Equilibria are saved during the scenario simulations and available for additional evaluation of the stability and for shape optimization studies. We have explored H-to-L and L-to-H transitions, beta (normalized stored energy) and l_i collapses, and controller-induced disturbances by suspending control and allowing the plasma to drift vertically unstable.

2. Scenario simulation

The simulated discharge parameters shown in Fig 1. are consistent with achieving Q=10 ($P_{fusion}/P_{external}$) performance, Fig. 1a, at P_{fusion}=400MW for 5.3T operation. In Fig. 2a, we show the shape evolution obtained using forward control with the JTC-2001, VS1 control of the six reference gaps shown to maintain the plasma boundary during discharge evolution. This ITER 15MA scenario starts with the large-bore shape on the outside limiter. The plasma is limited on the outside wall until about t=13.14s and then diverts with the shape rapidly mapping the strike points into the divertor region. The gap variations corresponding to this controlled shape evolution are show in Fig. 2b for the current ramp up phase and burn initiation. The early large variation in gaps is just the circular plasma coming off the limiter and being elongated near the top during the current ramp. In Figs 2b and 2c, we show the CS and PF coil currents obtained in this 700s simulation and note that all coil currents are within their allowed limits. We are just now beginning to study techniques for stable ramp down of ITER plasmas. For this simulation, we model an H-mode ramp down with density and auxiliary heating decreased during the plasma current ramp to limit the rise in l_i and γ_{VST}.

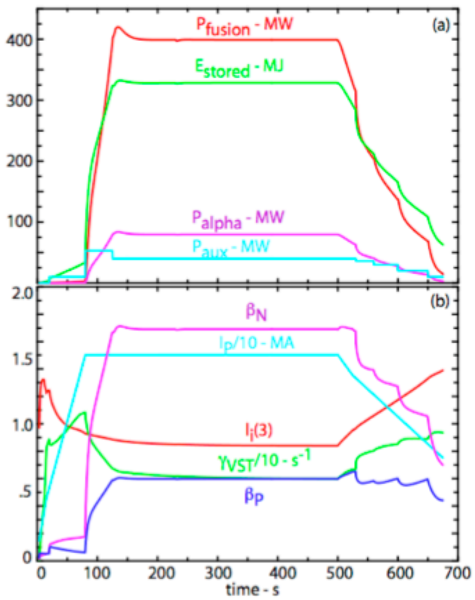

Fig. 1 15MA scenario with early heating: (a) Fusion power, stored energy, alpha heating and auxiliary power and (b) plasma current, I_P, normalized, β_N, and poloidal, β_P, beta, internal inductance, $l_i(3)$, and vertial stability growth rate, γ_{VST}.

Using a vertical stability calculation we show the resulting growth rates γ_{VST} in Fig. 1b along with the variation in internal inductance, $l_i(3)$, normalized stored energy and plasma current. The controller successfully maintains both the plasma shape and vertical position in this simulation. An issue with ITER scenarios is the limited

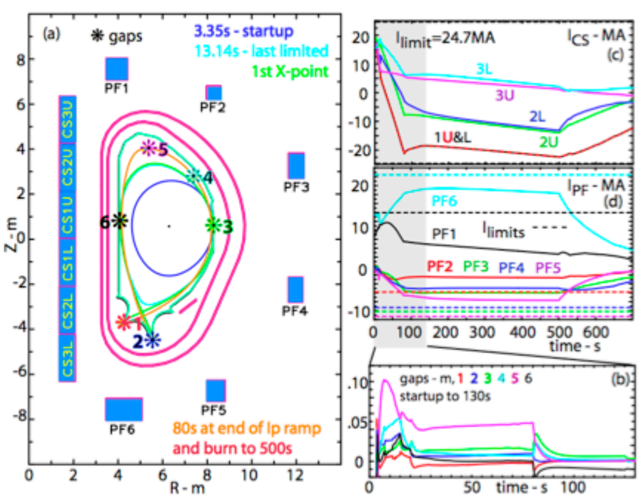

Fig. 2 (a) ITER coil geometry and plasma shape evolution controlled by gaps, (b) gap variations in controlling shape, (c) central solenoid currents and (d) poloidal field coil currents with allowed current limits.

volt-second capability for ramping up to full plasma current. In this scenario, 10MW of early auxiliary heating power, P_{aux} in Fig. 1a, is used to alleviate the demands on the CS1 coils, Fig 2c, that provide the Ohmic plasma current drive.

In these simulations, we are use a gyro-Bohm-based thermal transport model originally described by Tang [5] and Coppi [6]. This model was previously used for simulations of TFTR [7] and is now being validated with data from experiments on DIII-D. Impurity content and density profile in the simulations are prescribed except for the alpha particle density that is obtained from a rate equation with particle diffusion to limit the helium ash confinement. At current flattop, 80s, 53MW of auxiliary heating is applied to raise the stored energy to 340MJ (Q ~ 10) and the transport model parameters are modified to generate an H-mode-like barrier. The density profile is also broadened to simulate H-mode conditions. With these profiles, an edge-peaked bootstrap current is generated from a neo-classical model consistent with experimental observations. The edge transport conditions produce a 3.5keV pedestal during the burn conditions for the parameters in this simulation. We show typical profiles of density, temperature and toroidal current in Fig. 3 for both the L-mode current ramp at 50s the H-mode burn at 300s.

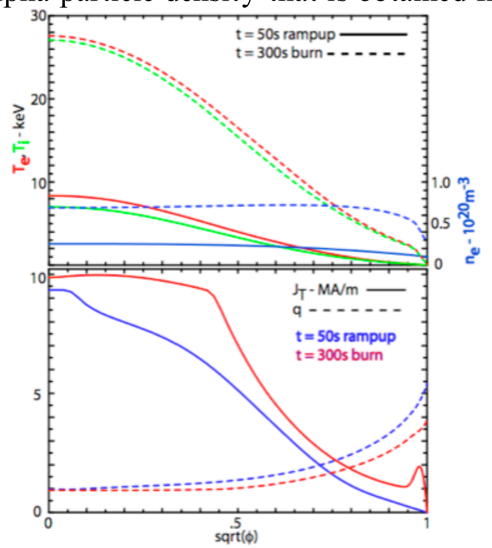

Fig. 3 Simulation profiles at 50s during rampup and at 300s in burn for (a) electron and ion temperatures from transport model and prescribed electron density and (b) total toroidal current density and resulting q profile with sawtooth model.

3. Stability modeling

We model controller sensitivity using various perturbations during the transport evolution. Of particular interest is an H-to-L back transition that experimentally might be stimulated by an interruption in the auxiliary heating power during the plasma current flattop and burn. In these simulations, we generate this event by turning off the auxiliary heating power and resetting the density profile and transport parameters back to L-mode-like conditions (as used during the current ramp). The rapid change in stored energy causes the plasma to move both radially and vertically. In the simulation shown here, Fig. 4, the controller is capable of regaining control as the plasma moved

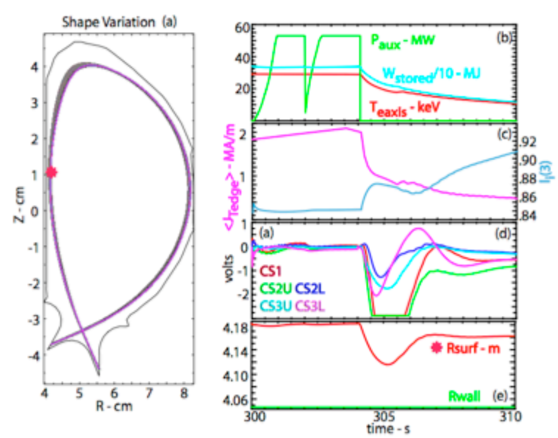

Fig. 4 (a) plasma shape evolution during H-to-L transition, (b) auxiliary heating turned off with stored endrgy and temperature decaying under L-mode model conditions, (c) edge bootstrap current decays and l_i increases, (d) CS control voltages temporarily saturate as control is regained, (e) plasma moves radially inward but controller keeps it from hitting inside wall.

towards the inside wall and upwards. While the controller voltages are saturated during this event, Fig. 4d, they eventually recover and control is regained. In the case shown here, $l_i(3) \sim .85$ at the start of the H-to-L transition, none of the PF or CS coil currents exceeded their allowed limits. In future work, we will explore worst-case back transitions to evaluate limitations of the controller and the coils.

We also study loss-of-control simulations where we suspend application of the calculated feedback control voltages and allow the plasma to move due to the vertical instability. After a desired vertical displacement is obtained, the controller voltages are re-applied to bring the plasma back to its equilibrium position. Results of such simulations indicate that a maximum displacement of ~ 3 to 4 cm can be tolerated in ITER.

4. Summary

We use free-boundary controller simulations to explore several aspects of ITER operating modes and evaluate limits and sensitivities of systems. We find that the ITER controller generally maintains the plasma shape and position during the current ramp up and flattop burn phases of the simulation. In these simulations, coil current limits during ramp up require the use of early heating avoid exceeding central solenoid current limits. During the flattop current and burn simulations, plasma control in the presence of H-mode peaked edge current profiles is successfully obtained. Most simulations show operating scenarios that push coil currents close to various PF coil limits. More work is needed to assess the margin of safety for these coils with the given controller. The ITER controller, however, does maintain control over the 700s discharge evolution and keeps the plasma off the central limiting surfaces during H-to-L back transitions at moderate l_i during the plasma burn. Worst-case events are currently under evaluation. Future work involves continued benchmarking with experimental data from DIII-D and development of satisfactory current ramp-down scenarios.

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References

[1] J.A. Crotinger, *et al.*, LLNL Report UCRL-ID-126284, 1997 available from NTIS #PB2005-102154

[2] T.A. Casper, *et al.*, Fus. Eng. Design **83** (2008) 552-556.

[3] T.C. Luce, *et al.* this conference

[4] G.L. Jackson, *et al.*, submitted to Nucl. Fusion.

[5] W.M. Tang, Nucl. Fusion **26** (1986) 1605.

[6] B. Coppi, Comments on Plasma Phys. Control. Fusion **5** (1980) 261.

[7] S.C. Jardin, *et al.*, Nucl. Fusion **33** (1993) 371.